Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/807
POLAND
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

CRITICAL AND EXPONENTIAL ASSEMBLIES AT THE INSTITUTE OF NUCLEAR RESEARCH, ŚWIERK

Collective work
Institute of Nuclear Research, Świerk, Poland

Since a few years ago activities to establish the experimental basis for research in the field of reactor physics and engineering have been carried out at the Nuclear Research Centre in Świerk. In 1963 three main research facilities have been put into operation : a swimming-pool type low-power reactor /MARYLA/, a graphite- and water - moderated critical assembly /ANNA/ and a graphite subcritical assembly /HELENA/.

A short technical description, basic physical parameters, and a survey of hitherto performed works at these facilities is given.

The programme of further research is also discussed. It includes:

1/ Reactor statics: buckling measurements for various geometries and lattices, micro- and macro- flux- distribution measurements, neutron temperature distribution measurements, application of the substitution method to reactor lattice investigations.

2/ Reactor kinetics: pile oscillator measurements, measurements of reactor parameters by noise methods, as well as with the aid of pulse neutron sources.

3/ A series of critical experiments concerning:

a/ the high-flux, graphite-water moderated reactor being designed at present,
b/ the medium-flux, light water moderated, 5 MW research reactor,
c/ the new core-type for the EWA reactor /in order to increase reactor power up to 10 MW/.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/808
SWEDEN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Fuel Handling Equipment for the Ågesta D_2O Moderated Pressure Vessel Reactor

G. Granelli [1], G. Fagerlund [2], G. Fröman [1]

The subject of the report is to give some aspects of the philosophy and the experience of design, manufacture and tests.

Certain basic parameters were stated for the design. The fuel handling has to be carried out under gastight and radiation protected conditions, the fuel handling machine and the fuel storage must be provided with an effective cooling system for all conceivable running conditions, the manoeuvering system must be fool-proof and finally, the D_2O condensate in the charge machine has to be collected and prevented from mixing with H_2O.

In order to make the design as simple as possible, it was decided to use a joint charge pipe for the four closest element positions. The charge pipe also contains the control rod which is serviced by the charge machine. The fuel elements hang from hooks below the reactor lid and are handled by means of a fork-grab located in the fuelling machine.

The fuel store is in the form of tanks, which are normally kept below atmospheric pressure to prevent activity leakage into the reactor hall. During fuelling, which is carried out in the "off-load"condition, the charge machine is connected, gastight, to the charge pipe in the reactor as well as to those in the fuel store.

The D_2O in the system is collected by draining the cooling gas. The condensate from the dryer is led to a special condenser tank. Any mixing of H_2O

1) AB Atomenergi, Stockholm
2) Nydqvist & Holm AB, Trollhättan

is prevented by thoroughly exhausting the charging pipe and fuelling machine before any charging is carried out.

Each operation is started and stopped manually. To make the system foolproof interlocks are used and manual overriding is provided on all basic operations.

The request for limiting the choice of construction materials and the stipulation for high cleanliness during manufacture has raised certain complications. The wish to keep down the dimensions has also necessitated designs which have caused certain manufacturing problems.

After necessary corrections during the final erection and tests it was shown that all functions corresponded well with the estimated design performance.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/809
SWEDEN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Design, Development and Manufacture of Control Rod Drives for Heavy Water Power Reactors

S. Ericsson [1], B. Ahlnäs [1]

The control rod drives of a nuclear reactor must be regarded as an auxiliary equipment and consequently their design must be adapted to the design of the complete core as optimized from an economic, heat transmission and reactor physics point of view. This leads to certain requirements with regard to location, space allowable etc. which it is the task of the designer to fulfil. On the other hand it might be impossible to satisfy the theoretically desired properties. Thus it is important that simultaneous and close collaboration is maintained during the design of the core and control rod drives.

In addition, for a control rod drive, the requirment for functional safety is extremely high. For heavy water reactors there is also the special problem due to health hazards in connection with the tritium content of the heavy water. This must be taken into account when planning for the exchange of drives in the reactor, for service work on the mechanisms etc.

In the paper a description is given of the experience which has been gained in the Swedish work on D_2O moderated and -cooled pressure vessel reactors, one being the Ågesta reactor, which is now in operation, and the other the Marviken reactor, which is in an advanced design stage and is due to be critical in 1968.

In both these reactors the absorber rods are tubular and the absorbing alloy, silver, indium and cadmium, is canned on the inside and outside with stainless steel. In the boundary region the absorbing alloy and the canning material

1) ASEA, Västerås

are metallurgically bonded to each other. The development work and the manufacture of the control rods is dealt with in the paper.

The drive mechanisms are of the same principal design in Ågesta and Marviken. The differences in location and environment conditions, however, make a number of modifications necessary for Marviken. In the paper a description is given of the design of the drive mechanisms, of the results from the extensive development work which was performed for Ågesta and of the planned prototype tests for Marviken.

A number of special problems had to be solved, during the manufacture of the control rod drives for Ågesta. The most important of these as well as the first operational experience are touched upon in the paper.

Finally some conceivable types of control rod drives for developed heavy water power reactors are mentioned. One solution, to which a great deal of design and experimental work has been devoted, is a drive for location below the reactor core in the pressure vessel bottom. A short description of the work on this design is given.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/810
SWEDEN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Design and Manufacture of the Reactor Pressure Vessels for the Ågesta and Marviken Power Stations and some Future Developments

O. Hellström [1)]

The Swedish nuclear power program has from the beginning been concentrated upon heavy water and natural uranium reactors. After some studies it was decided to make the Ågesta reactor of the pressure vessel design type. From a general point of view the Ågesta reactor might be regarded as a prototype for two attractive outlines of pressure vessel unit developments for competitive nuclear power, namely

1) the uniform lattice reactor (homogenized), PHWR
2) the natural circulation direct cycle boiling reactor equipped with facilities for nuclear superheating, BHWR.

The Marviken Power Station is of the second type and is designed for 140 Mw elect. for saturated steam operation and 200 Mw elect. with nuclear superheat operation. The pressure vessel for this plant is designed and the manufacture has started.

The paper describes the design and manufacture of the Ågesta and the Marviken reactor pressure vessels. Code requirements, types of material and properties and special design calculations of interesting parts are discussed. Further, the special manufacture problems are delt with which is due to the weight, geometric sizes and high tolerance requirements of these pressure vessels. The shop facilities must be sufficient to take care of these special characteristics. As for the increase in geometric size it can be postulated that with use of better materials and adjusted fabrication methods the size problems can be taken proper care of during the next 5 - 10 years. For plant sizes considerably above 500 - 600 Mw elect. it is likely that the pressure vessels must be prefabricated in the shop and site assembled by using special technique.

---

1) Uddeholms AB, Degerfors

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/811
UNITED ARAB REPUBLIC
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

AN EXPERIMENTAL STUDY OF Sr^{90} CONTAINED IN MARINE ANIMALS FOLLOWING POSSIBLE RELEASE OF RADIOACTIVE WASTE IN SEA WATER

H.A.F. Gohar, S.E.E. Hashish,
A.A. Hassan and K. Moulikhia

Nerita forskalii Recluz: a gastroped belonging to the family Neritides is found in great abundance along the shore of the Red Sea as well as the Gulfs of Suez and Aqcaba. It is of ubiquitous occurrence throughout the region of the Indian ocean and the warmer parts of other oceans. It contains comparatively high concentration of Sr, and could be kept in glass aquaria indoors. For these reasons the gastroped was used for the investigations.

The study comprised the following:

1 - The ionic content in the animal shells determined flame-photometrically were found 33.1 ± 0.4 (26), 183 ± 5.8 (27), 15.6 ± 0.4 (25) and 673 ± 8.3 (27) as mg/100 g dry shell ± s;d. (No. of experiments) for Ca^{++}, Sr^{++}, K^{+} and Na^{+} respectively. The corresponding values in the soft tissues were 750 ± 10 (9), nil, 461 ± 39 (9) and 1438 ± 285 (9).

2 - The average values for Sr in shells as mg/100 g dry shell of animals weighing less than 2 g and more than 2 g were found 222 ± 8.4 (7) and 169.5 ± 4.2 (20) respectively, i.e. young animals contain more Sr in their shells than older ones.

3 - The ionic content in the animals was statistically the same for those collected from Al-Ghardagua and Suez.

4 - The ionic content in sea water as mg/litre was 588, 72, 643 and 13.1 x 1000 for Ca, Sr, K and Na respectively.

5 - The Sr^{90} content in 100 g dry animal shell was found 3357 ± 107 (12) in terms of u.u.c. or 101.4 ± 3.3 (12) in terms of Sr units, and in one litre sea water 5.4 ± 0.1 (6) u.u.c., and 9.0 ± 0.2 (6) Sr units.

6 - The $T_{\frac{1}{2}}$ for the uptake of Sr^{90} by the animal shells from sea water containing 90 u.c./litre during a period of one month increased progressively from 1.4 to 37.5 days. The fraction exchange of Sr in the shells increased for the same period from 0.0036 to 0.0741.

7 - The $T_{\frac{1}{2}}$ for the uptake of Sr^{90} by the animal soft tissues from sea water containing 40 u.c./litre during a period of one month increased progressively from 3.4 to 15.0 days. When the external activity was raised to 90 and 190 u.c./litre the $T_{\frac{1}{2}}$ during the period 16-28 days of uptake were found 16.3 and 18.0 days respectively.

8 - The ratios $\frac{Sr^{90}/\text{litre sea water}}{Sr^{90}/100\ g \text{ dry shell or soft tissues}}$ = RW/SL or RW/ST were determined. These could be considered as valuable indices for the calculations of Sr^{90} content in sea water by determining the Sr^{90} content in either the shell or the soft tissue for a given period of uptake.

9 - The $T_{\frac{1}{2}}$ for the release process of Sr^{90} from the soft tissues was found to be 1.25 days for the first day and 30 days for the following 6 days. Sr^{90} was not detectably released from the shells.

10- The kinetics and the nature of the processes involved in the uptake and release of Sr for the gastroped in the light of the results obtained are discussed.

This work has been carried out as a research project (No. 58) between the UAR, AEA and IAEA. The authors are indebted to both institutes for technical and financial supports.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/812
UNITED ARAB REPUBLIC
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

EVALUATION OF DIFFERENTIAL DOSE DISTRIBUTION IN HUMAN PHANTOM USING S^{32} (n,p) P^{32} REACTION IN HAIR

K.A. Mahmoud, A.N. Hanna and E.A. Amin
Radiation Protection and Civil Defense Department,
Atomic Energy Establishment, U.A.R.

In certain reactor accidents, human subjects are sometimes exposed to neutron beams which deliver unknown neutron dose to different human organs. The S^{32} (n,p) P^{32} reaction due to neutrons of energy greater than 2 MeV was used as basis for neutron dose measurements. The neutron source was one of the horizontal neutron beams of the U.A.R. water-water cooled reactor and the sulphur present in human hair (30-50% dry weight) was the target material for neutron detection. Samples of human hair were placed at specific points of standard man phantom, which represent active bone marrow and gonads. The bone marrow was represented by the Ribs 1 to 4, 5 to 8, 9 to 12, Cranium and mandible Scapulae, humeral heads and clavicles - Sternum - Sacrum - Os Coxae - Femoral head and neck - Cervical Vertebrae - Thoracic Vertebrae 1 to 4, 5 to 8, 9 to 12 and Lumbar Vertebrae. The gonads were represented by left and right ovaries for female, and scrotum for male. The standard man was designed and constructed locally from a real human skeleton, filled with sherezine wax. Lungs were assembled in the form of two plastic bags filled with wet sawdust while intestines were replaced by strips of rubber foam soaked in paraffin. Samples of the same hair were also placed at certain points on the top of the phantom skull. The irradiation time and the positioning of the phantom in front of the neutron beam were chosen in a way to represent typical conditions of irradiation of occupational workers in case of accidents or negligence of protectional signs when neutron horizontal beams are left open during reactor operation. The sulphur and phosphorus content of these hair samples inserted in the phantom, were assumed to represent the case in vivo.

The experimental dose and dose rate data received by bone marrow, gonads and hair of the skull were used for derivation of certain parameters correlating the dose or dose rate received by the skull hair to that received by gonads and bone marrow for different irradiation conditioning and for different phantom positions. These parameters were tabulated in a certain form which could facilitate the assessment of gonad or bone marrow dose received by man after accidental neutron exposure.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/813
UNITED ARAB REPUBLIC
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

REVIEW OF THE STUDIES UNDERTAKEN IN THE U.A.R. FOR EVENTUAL DISPOSAL OF LOW ACTIVITY WASTES OF STRONTIUM, RUTHENIUM, CAESIUM AND CERIUM INTO THE GROUND

M.A. El-Guebeily, M.Y. Farah, A.E. Sinbel, M.S. El-Alfy, S. Shakour, M. Fouad and G. El-Shobaky

The problem of disposal of radioactive wastes constitutes one of the important aspects influencing the development of the peaceful application of atomic energy, particularly in relation to its impact on the economics of nuclear power. The present paper summarises the fundamental data obtained in the U.A.R. Atomic Energy Establishment precluding the feasibility of ground disposal of such wastes in soils and on minerals from the country. Radionuclides were exemplified by radioactive ruthenium, caesium, strontium and cerium; while ground matrix was tested on Mokattam and Sinai clays, sands from Reactor Site at Inshas, Nile valley phosphorites, zircon placers, Aswan hematites, etc... . These ores and soil minerals were studied for various sizes of their particles and over a wide range of pH values. The paper collects also the fundamental data for optimising the decontaminating efficiency of each alternative. Batch as well as column experiments were undertaken and complete analyses of soils and clays spectrographically as well as by X-ray powder diffraction are given.

A classification of minerals and radioelements in the order of efficiency of decontamination is given, as also calculation of exchange capacities. The effect of trace impurities of iron and calcium in enhancing uptake is also investigated.

A mixed bed is foreseen for decontaminating low active wastes of these radioelements. Further investigations are envisaged for other radioelements and other minerals to enable the determination of the final specifications of the mixed natural bed.

The physiochemical laws governing the disposal of other heavy radio-elements, particularly thorium and uranium, into the ground are presented in a second paper to this conference.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/814
UNITED ARAB REPUBLIC
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

UPON THE PHYSICOCHEMICAL LAWS GOVERNING THE DISPOSAL OF SOME HEAVY RADIOELEMENTS INTO THE GROUND

M.A. El-Guebeily and N.Z. Misak
Atomic Energy Establishment, U.A.R.

A series of studies have been previously undertaken in UAR.AEE to examine the feasibility and the characteristics of disposal of radioactive wastes of Sr, Ru, Cs and Ce into the ground. The present work aims at complementing these studies by data upon the behaviour of some heavy alpha emitting radioelements and in particular to examine the physicochemical laws governing their interaction with sand and clay minerals. For this purpose uranium and thorium were chosen in the forms of uranyl nitrate in HNO_3, uranyl carbonate complex in Na_2CO_3 solution and thorium nitrate in HNO_3; the concentration of U and Th being of the order of 10^{-3}M. As adsorbing materials, Inshas reactor site sand and an Egyptian vermiculite were used. Both batch and column techniques were applied in the uptake experiments. The percentage decontamination and the pH value were followed with time till after equilibrium. The effects of initial pH, grain size, temperature and concentration on the uptake process were investigated. The change of concentration of U and Th was followed spectrophotometrically.

The results obtained indicate that:

(a) Uranium in the anionic form is not taken up by sand and is only slightly absorbed by vermiculite.

(b) UO_2^{2+} and Th^{4+} are very slightly taken up by largest size sand.

(c) UO_2^{2+} and Th^{4+} uptake by finest and intermediate size sand, as observed from the kinetic curves, is rapid for relatively low initial pH values, but a second slow process appears where the pH of the solution exceeds the value of 2.

(d) The uptake of Th^{4+} by intermediate and finest vermiculite sizes is very rapid, in contradistinction to UO_2^{2+} for which equilibrium is attained in about 30 minutes. With largest size, the results are irreproducible.

(e) H^+ is taken up in parallel with UO_2^{2+} and Th^{4+}. The capacity for UO_2^{2+} and Th^{4+} increases with increases of initial pH and with decrease of particle size.

(f) The effect of temperature on uptake is minor in all cases.

In order to correlate and interpret the main uptake data, the following physicochemical treatment was tried:

1 - To evaluate the number of processes involved in the uptake phenomena, the equation of unimolecular equilibrium

$$-(k_1 + k_2) = \frac{1}{t} \ln\left(1 - \frac{x}{x_e}\right)$$

was used as an approximation, neglecting other influences. This equation has the form of McKay's equation

$$k = -\frac{ab}{a+b} \cdot \frac{1}{t} \ln\left(1 - \frac{x}{x_e}\right)$$

with difference however in the physical meaning of the constants. The number of processes obtained is in accordance with the kinetic curves on sand and the known properties of vermiculite.

2 - The evaluation of the equilibrium distribution coefficient

$$k_d = \frac{\% \text{ adsorbed}}{100\% - \% \text{ adsborbed}} \cdot \frac{V}{M}$$

at different acidities is in favour of an ion exchange adsorption mechanism.

3 - To evaluate the rate determining step, the particle diffusion equation

$$Q_t/Q_e = 6Q_o\sqrt{\frac{Dt}{\pi}} \;/\; r\,(Q_o - Q_e)$$

was applied and it was found that in all cases the main uptake processes are most probably particle diffusion controlled. The variation of the internal diffusion coefficient with pH is indicative of changes in the structure of the adsorbable ionic species.

4 - The data on sand may be described quantitatively by Langmuir isotherm

$$\frac{C}{x/m} = \frac{1}{k_1 k_2} + \frac{C}{k_2}$$

while those on vermiculite may be correlated by a Freundlich isotherm

$$\frac{x}{m} = kC^{\frac{1}{n}}$$

with more than one deflection, indicating the possible presence of different energy sites.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/815
AUSTRIA
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Measurement of High-dose Gamma Fluxes at Reactor Pulses and Strong Gamma Sources by the Conductivity of Cuprous Oxide

O.Bobleter und P.Vajda
Atominstitut der Österreichischen Hochschulen, Vienna

Cuprous oxide with a coarse crystalline structure was produced from copper plates (1 x 20 x 80 mm) by a special oxidation process. The specific electric conductivity of this material is not different from that of single crystals. In the case of fine crystalline specimens, the conductivity increases proportionately with the number of crystals per unit area. This behaviour is the result of higher concentration of crystal defects on the grain surfaces.

Under the influence of gamma and X-ray irridiation the conductivity is increased. The lower the initial conductivity of the material, the greater is the relative effect. The results thus obtained showed a linear dependence and proved coarse crystalline Cu_2O to be most suitable for high gamma flux determinations.

Immediately after the commencement of irridiation by a strong gamma source, a sharp decrease of the Cu_2O resistance is measured; after a few minutes the curve becomes less steep and turns into a linear section. When irradiation is interrupted, the first decrease recombines almost completely in a very short time, while the annealing of the defects characterized through the linear section of the curve, takes place very slowly and is therefore a long-term effect.

Reversible conductivity changes in the Cu_2O specimens were in fact obtained by short gamma pulses at the reactor (TRIGA Mark II).

Due to the inherent safety of the reactor, power pulses of up to 300 MW can be obtained. The half-width of the pulse is approximately 30 msec. The Cu_2O specimens were covered with cadmium and placed in positions where gamma dose-rates of several kr/h (30 - 70 r/pulse duration) were obtainable. The curve of the conductivity measured during the pulse is almost identical with that of the reactor power. It is possible with this comparatively simple Cu_2O conductivity measurement device, to measure the peak power of the reactor pulse and to transfer electronically the data obtained. As far as a reactor pulse can be taken for a simulation of a power excursion in a nuclear fuel plant, the above method of measuring the Cu_2O conductivity can be used as a monitoring system for nuclear power accidents.

Furthermore Cu_2O specimens can be used to measure high doses gamma fluxes from continuously radiating sources by putting the sample for a certain time in the place in question and subsequently measuring the conductivity. The mechanism of the conductivity and the influence of the radiation on the crystal defects is briefly discussed.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/816
UNITED ARAB REPUBLIC
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

ERECTION, TESTING AND OPERATION OF THE RADIOISOTOPE PRODUCTION FACILITY OF THE UAR AEE

M.A. El-Guebeily, E. Hallabe and M. El Garhy, et al.*

A general description of this project has been presented to the IAEA symposium on Programming and Utilisation of Research Reactors held in Vienna, October 1961. Construction of this facility has been completed in July 1962 and was followed by extensive testing till its commissioning on 23 December 1962, after which routine operation and distribution continued regularly till to date. This laboratory has been the result of the fruitful co-operation between the U.A.R. A.E.E. and the Norwegian firm NORATOM, under an Agreement of co-operation between U.A.R. A.E.E. and I.F.A. on the Peaceful Applications of Atomic Energy.

The radioisotope production laboratory includes facilities for the routine production of I^{131} (1c per batch in each of two boxes), Au^{198} (colloidal and Muller types, 5c each per batch), Na^{24} (50 mc per batch), K^{42} (2c per batch), P^{32} (500 mc per batch), S^{35} (500 mc per batch) and Cr^{51} (500 mc per batch). It also includes other general hot facilities for research and development as well as various facilities for control of production. The present report deals with the main lines of experience gained in erection, testing and operation of the radioisotope production laboratory, as well as with the main lines of development.

The construction stage gave valuable experience to various engineers and skilled workers in erection of hot cells and production boxes and ventilation, monitoring and radioactive waste systems. The main experience in this connection is the absolute necessity of intimate co-ordination and co-operation of the various engineering groups, both in planning and in realization.

On completion of construction, testing was carried out on two stages as follows:

(a) Testing of the different individual apparatus and equipment, followed by testing of the complete technological system, ventilation, waste, etc.

(b) Testing of the production processes themselves starting with inactive runs, followed by low activity production till full capacity tests. All medical products have been subjected to strict radiochemical, spectrochemical and biological control, and proved suitable for medical distribution.

Consequently, regular production has been carried out, which satisfied all the nedds of the U.A.R. from these isotopes, as well as the demands of the Middle East Regional Radioisotope Training Centre for the Arab Countries. Although the laboratory production capacity is much higher than the actual production during this initial running-in-period, yet the distribution dates indicate a general trend of increase in the application of these isotopes in the U.A.R.

In parallel with the regular production, radiography sources of Co^{60} and Ir^{192} have been produced on demand and a development programme has been undertaken in co-operation with Dr. Micheev. For production of colloidal solutions of Ag^{111}, Zr $OHP^{32}O_4$, and Y^{90} silicate, as well as Tc^{99} generator, a new facility is at present being planned for regular production of these materials, profiting of the experience of the present designs. A wide programme for production of a variety of unprocessed isotopes and radiation sources as well as of labelled compounds is under consideration.

---

*a) Many engineers and radiochemists, whose names will be mentioned in the text of the main report, have contributed graciously in various lines at different stages.

b) Engineers Axelsen and Hoye represented NORATOM in erection and testing.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/817
BRAZIL
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

DETERMINATION OF THE PLUTONIUM 240 CONTENT IN PLUTONIUM BY MEANS OF NUCLEAR EMULSION

by

H.G. de Carvalho and A.G. da Silva
Instituto de Engenharia Nuclear, Guanabara,
Brasil

Plutonium isotope composition is an important parameter in a plutonium fuel cycle in thermal nuclear reactors. This paper describes an inexpensive method for the determination of small quantities of Pu 240 in concentrations above 0.1% in the plutonium fuel, which may be used when a mass spectrograph is not available. The method is applied to any binary mixture of transuranium elements with very different spontaneous fission half-lives. The half-life for spontaneous fission of even-even nuclides is in general smaller than that of even-odd isotopes of a particular element. The half-life for spontaneous fission of Pu 239 is $4,5 \times 10^4$ times longer than that of Pu 240. This phenomenon in case of binary mixture of isotopes, allows a method for determination of the composition provided that for a determined mass of the element the spontaneous fission activity can be measured. This activity may be measured by means of the neutron emission from the spontaneous fission or by measuring the spontaneous fission activity.

By means of a technique for discrimination development of nuclear emulsion which allows a perfect discrimination between fission and alpha particle tracks in nuclear emulsion (Ilford K0 type), it is possible to record only the spontaneous fission tracks in the presence of intense alpha particle background. A weighed amount of chemical pure plutonium, converted to nitrate, is transformed into a thin film of inert plutonium

oxide by means of the technique for preparation of thin films with alcohol-ether-colodium nitrate solutions and then heated in a oven at 600°C. The film suport is made of stainless steel or hard-melting glass.

The nuclear emulsion is put in contact with the thin film oxide according to the Pu 240 content for a period of up to 20 days. After processing and scanning, it possible to measure activities so low as to produce 70 fission tracks/cm^2 in 20 days. Such an activity corresponds to an amount of 0.0001 mg of Pu 240. In the plutonium oxide film of 0.1 mg/cm^2 it corresponds to the presence of 0.1% of Pu 240.

By comparison with a standard fission neutron source it is also possible to ascertain the Pu 240 content by means of B-10 loaded nuclear emulsion in samples which contain a mass of the order of 10 mg or more of Pu 240.

The isotopic content of the plutonium produced in a U-238 U-235 reactor depends on the irradiation time, neutron flux, conversion ration and fuel enrichement, for example a 0.715 per cent burnup of the total uranium yields proportion of plutonium isotopes 239, 240, 241 and 242 of 65.1, 15.3, 16.1, 3.5 per cent. The half-lives for spontaneous fission of Pu 242 and Pu 240 are, respectivelly 7 x 10^{10} and 1.25 x 10^{11} years. In cases of high burnup, when the content of Pu 242 is considerably high the additional measurement of the gamma activities is necessary to obtain the ratio Pu 240 to Pu 242, since their spontaneous fission activities were measured together.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/818
ISRAEL
February 1964
Submitted in:
ENGLISH and FRENCH

Confidential until official release during Conference

Abstract

ELECTROCHEMICAL REDUCTION OF DILUTE NITRIC ACID

J.A. Epstein, I. Levin and S. Raviv

Israel Atomic Energy Commission

Dilute nitric acid is a major constituent of many radioactive liquid waste systems. The methods currently used for its disposal, such as neutralisation or reduction with formaldehyde, suffer from several disadvantages.

Electrochemical reduction of nitric acid generally gives rise to undesirable products, such as ammonia, which interfere in subsequent processing of the waste. An attempt to find an electrochemical reduction process which would reduce nitric acid to gaseous products, and would not require the addition of foreign ions to the system, led to an investigation of the behaviour of a graphite cathode in dilute (0.5 - 3N) nitric acid.

Current-voltage curves, determined by a potentiostatic method, were measured at different acid concentrations, at different temperatures, and in the presence of urea and sodium nitrite. Electrochemical reduction of nitric acid takes place at positive potentials, the products of reduction being mainly gaseous oxides of nitrogen. The presence of carbon dioxide has also been established among the products. The rate of the electrochemical reaction increases rapidly with increase in nitric acid concentration. There is evidence to show that the rate varies linearly with the concentration of nonionised nitric acid. At low acid concentrations, presence of nitrite increases the rate, while at high acid concentrations the addition of nitrite reduces the rate of the electrochemical reaction.

The rate is also reduced in the presence of urea. The results indicate that one of the electrochemical reactions involved is the reduction of nitrous acid, which

may be formed by the chemical reduction of the nitric acid by the graphite.

One of the possible reaction schemes is the following:

1. Chemical reduction

$$HNO_3 + C \rightarrow CO + HNO_2$$

$$2HNO_3 + C \rightarrow CO_2 + 2HNO_2$$

2. Electrochemical reduction

$$HNO_2 + e + H^+ \rightarrow NO + H_2O$$

3. Oxidation

$$CO + O \rightarrow CO_2$$

$$NO + O \rightarrow NO_2$$

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/819
ISRAEL
February 1964
Submitted in:
ENGLISH and FRENCH

Confidential until official release during Conference

Abstract

THE CHEMISTRY OF THE TRIVALENT ACTINIDES IN AQUEOUS SOLUTIONS AND THEIR RECOVERY

Y. Marcus, M. Givon and M. Shiloh

Israel Atomic Energy Commission

The irradiation of plutonium in a reactor, as fuel or target, eventually produces actinide elements, besides the fission products. The former can be irradiated in turn to yield still higher actinides, which have some practical uses - curium as a heat source in satellites, californium as a neutron source, and others. The recovery and purification of these materials necessitates a familiarity with their chemistry.

The transplutonium elements occur mainly in the trivalent state, and have chemical properties similar to those of the lanthanides. However, the chemistry of both groups in aqueous solutions useful for separation processes has only recently been studied in detail. Ion exchange and solvent extraction processes using acid phosphate esters, hydroxy-carboxylic acids and concentrated chloride and nitrate solutions play important roles in the separation of the actinides, mutually and from other elements.

A survey of the chemistry of the actinides in these and related media shows how the properties of these complex systems contribute to the separation and recovery processes.

The separation of actinides by elution with, e.g. alpha-hydroxyiso-butyric acid, from a cation exchanger, parallels the separation of the lanthanides. This process, indeed, was the only one capable of allowing the chemical identification of the heaviest actinides produced. The separation factors for neighboring elements are moderately high (1.6). These elements form complexes with up to four bidentate chelating ligands, and the anionic complexes formed are octo-coordinated. The relative stability of the actinide and lanthanide complexes with these and related ligands may

be correlated with properties such as the crystal ionic radii. Ligand field effects seem to be unimportant.

A competitive process for mutual separation of the actinides is their extraction (e.g. in reversed phase chromatography) by an acidic phosphate ester, such as di-(2-ethylhexyl) phosphate. The separation factors (2.5) are even larger than in the above process. The complexes formed generally have three dimeric ester ligands, but complications occur in certain systems. The synergistic action of reagents such as T.B.P. can enhance distribution coefficients and mutual separability. These actinide complexes have apparently coordination numbers of either six or eight.

The above processes cannot separate the actinides from the lanthanides present as fission products in process solutions. This can be accomplished by anion exchange or amine extraction from concentrated lithium chloride solutions at low acidity. This indicates that anionic complexes are formed, but spectrophotometric measurements in our laboratory on the lighter trivalent actinides, uranium to americium, show that only weak cationic complexes are active spectroscopically, as shown by the appearance of new U.V. peaks.

Good anion exchange separation of americium and curium may be obtained with lithium nitrate rather than chloride. The behaviour of the heavier actinides has also been compared in this laboratory with the unusual even-odd atomic number effect shown by the lanthanides, and some information on the electronic state of these ions has been obtained.

In conclusion, although some separation processes utilise the different oxidation states of the actinides to obtain efficient separation (e.g. of Am(VI) and Bk(IV)), the key to their recovery still lies in the knowledge of the behaviour of the trivalent state in aqueous solutions.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/820
INDIA
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

DETERMINATION OF RATIO OF U-235 AND U-238 IN URANIUM

A. Ramamoorthy and A.K. Ganguly
Atomic Energy Establishment Trombay

A method to determine the isotopic ratio of U-235 and U-238 in uranium samples by measuring the build-up of their beta active daughters, Th-231 and Th-234, is considered. These daughter products are quantitatively separated in the carrier-free state in an anion exchange column and estimated using a flow counter. The accuracy of the method depends essentially on the total beta counts recorded for estimation. The results presented in the paper were computed from experiments recording about million counts in each case. These observations are then used to establish half-lives of the beta active daughters of pure uranium. From this study isotopic ratios of U-235 and U-238 are determined in the uranium sample.

A number of natural uranium, depleted and enriched uranium samples from different countries were analysed using this method and the results are presented. The paper discusses the merits of the method in comparison with other methods available for isotopic estimation of uranium.

Systematic variations in the ratios of U-235 and U-238 in natural uranium samples from different ores and uranium concentrates have been observed in these experiments.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/821
PAKISTAN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

THE SOLVENT EXTRACTION OF THORIUM FROM SULPHURIC ACID SOLUTIONS BY AMINES

M.A. Awwal
Pakistan Atomic Energy Commission

Amines offer certain advantages as solvents compared with alkyl phosphates in processing irradiated fuels due to the radiation susceptibility of the latter. While extensive studies have been made of the potential of amines in uranium ore processing in sulphuric acid media, less work has been done on the possibility of employing amines as extractants in irradiated fuel reprocessing especially in the Th-U^{233} cycle. Basic chemical studies associated with this possible processing system are reported.

Primary, secondary and tertiary amines have been investigated. Use of primary amines was found to be impracticable whereas secondary or tertiary amines are promising.

Two secondary amines, di-n-octylamine and di (tridecyl) amine as well as two tertiary amines, tri-n-octylamine and methyl-n-didecylamine, were studied in detail. Both the secondary and tertiary amines have shown suitable extraction properties for both uranium and thorium from sulphate media especially in the range 10^{-2} to $10^{-1} NH_2SO_4$. The secondary amines show a preferential extraction of thorium while the tertiary amines preferentially extract uranium. The secondary amine sulphates at a concentration of 0.1 M have a partition coefficient for the extraction of thorium which decreases from a value in excess of 10^3 for acid concentrations of $10^{-2} NH_2SO_4$ to approximately unity at aqueous acid concentrations of 0.5N. Under similar conditions uranium gives similar results but reduced in magnitude by a factor of four. With 0.1 M tertiary amines the partition coefficient for the extraction of thorium decreases from a value of approximately 5 at $10^{-2} NH_2SO_4$, to a value of approximately 0.5 at 0.5 NH_2SO_4. Under the same conditions uranium gives a partition coefficient V's acid concentration

curve of similar shape but with partition coefficients increased by a factor of 10^2.

Thus while the secondary amines provide the higher partition coefficients for the extraction of uranium and thorium, the tertiary amines provide the greatest difference between the values for the extraction of uranium and thorium under similar conditions. Because of this although the secondary amines would appear attractive to extract both the uranium and thorium from a feed solution the tertiary amine solvents offer better subsequent thorium and uranium separation by selective stripping of the solvent.

In partition coefficient V's amine sulphate concentration studies a greater dependence and hence flexibility in the selection of extraction conditions was found for the tertiary amines. These results also pointed to at least two different extracting species although paper electrophoresis experiments which clearly showed the presence of anionic species failed to confirm the existence of multiple species.

Thorium loading studies show a far greater falling off in extraction with secondary amines for thorium concentrations above 0.01 M than is the case with tertiary amines so that whereas secondary amine partition coefficients are 100 times higher than tertiary amine coefficients at thorium concentrations of 0.005 M they are of almost equal value at thorium concentrations of 0.05 M.

The greater differences in the partition coefficients for thorium and uranium with tertiary amines together with the greater dependence on amine concentration and similar overall partition coefficients at the higher thorium concentrations indicates that the tertiary amines provide the better basis for a uranium and thorium isolation and separation system in spite of the apparently greater attraction of higher extraction coefficients for secondary amines in less comprehensive tracer level studies.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/822
PAKISTAN
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

MEASUREMENT OF LATTICE PARAMETERS USING A SINGLE FUEL CHANNEL[1/]

Dr. S.A. Durrani
Atomic Energy Centre, Lahore

Zink and Rodeback have shown that it is possible to obtain values of thermal utilization, E, and resonance escape probability, p, for a heterogeneous thermal reactor, using a single fuel element embedded in the moderator. This method was tested by using a single uranium fuel channel, 99" long, placed along the axis of a cylindrical assembly of graphite constituting an infinite lattice.

A neutron generator was used to provide a flux of 14 MeV neutrons which were first slowed down by a thermal column. At roughly 10 cm intervals along the axis, graphite rods were inserted radially into the assembly, carrying In foils which measured the In resonance flux φ_{in} at 2 cm intervals along each radius.

In such a case $p_{\infty in} = \int_a^\infty \varphi_{in}\, \gamma\, d\gamma \,/ \int \varphi_{oin}\, \gamma\, d\gamma$ where φ_o is the flux with a non-capturing fuel element, and a is radius of fuel element in an infinite lattice. A dummy (lead) fuel element was substituted to simulate a non-capturing fuel element. Also, fuel elements from UKAEA, which had been used in an exponential experiment giving a known value of buckling, B^2, in a full assembly, were used for calibration purposes.

In the paper, the experimental results of these measurements are described. The thermal utilization was obtained correct to within 1/2%, but resonance escape probability was fouled by streaming and by the fact that the fuel channel was not truly infinite but had an extra source term.

This is an economical and simple method of obtaining parameters which are of great importance for power reactor design and is a logical reduction of the exponential method. It is intended to continue this experiment at the Atomic Energy Centre, Lahore, using a neutron generator and an assembly with water as the moderator.

1/ Work performed at U.C.P. Laboratories London, in collaboration with R. Berenbaum and E.W. Etherington.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/823
UNITED ARAB REPUBLIC
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

ON THE BEHAVIOUR OF NITROSYLRUTHENIUM NITRATO COMPLEXES IN TBP-HNO_3 SYSTEM

M.A. El-Guebeily, E. Hallaba and Reda A.I. Azzam

This work was undertaken to throw some light on the solvent extraction behaviour of the nitrato complexes of nitrosylruthenium in TBP-HNO_3 system. Both tracer (10^{-9}M) and macroscopic (10^{-3}M) concentrations of ruthenium have been investigated. For tracer work, the complexes were prepared by Martin's method while for macroscopic one, Fletcher's procedure was employed.

A comparative study of both preparates at 10^{-3}M Ru concentration, was undertaken spectrophotometrically. This has revealed, among other things, that Fletcher's procedure gives directly the nitrato complexes while Martin's method leads initially to RuO_4 in nitric acid which undergoes a series of successive reductions leading to the nitrato complexes after 17 hours. All tracer work was carried out after suitable aging time.

The extraction studies started with the system TBP-H_2O-HNO_3 without Ru and showed that the extractable species has the form TBP $(H_2O)_x$ HNO_3, where x was found to be equal to 1 and 4 depending on the method of evaluation, with more evidence in favour of 4. The equilibrium constant K for the formation of this product was calculated.

On the other hand, the extraction behaviour of nitrosylruthenium nitrato complexes at tracer and macroscopic concentrations was followed radiometrically. Measurements were carried out in equilibrium conditions. The results indicate the following:

(1) The variation of D_{Ru} with HNO_3 concentration has the same general feature for concentrations of TBP between 15 and 60% and for a very wide range of Ru concentrations (10^{-9} to 1.65×10^{-3}), irrespective of the method

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/824
TUNISIA
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

Nuclear Plants for Power and Saline Water Convertion

Tunisian Atomic Energy Commission

-=-=-=-=-=-=-=-=-

Although developing countries economy is based on agriculture, big efforts are done for establishing industries. So water energy are in most cases the basic requirements of economic development.

Many problems arise by shortage of water in areas of some countries : migration of population from one region to another, economic unbalance due to industry set up in prefered areas....

Even in industrial areas and great centers of population, water requirements of industrial and domestic purposes are increasing and in several cases water supply of these regions by pipes is not the best solution ; then, water conversion plants are to be considered among alternatives water supply.

Among energy sources required for water convertion plants, nuclear energy may be considered. Nuclear source energy may offer advantages especially for dualpurpose plant.

Many problems of course have to be solved but reactors and saline water conversion plants technology at present time has no great difficulties to be solved.

Generally, it is possible to foresse :

- types of reactors to be adapted for dual-purpose.
- processes for desalination which have the great experience.
- the problems of connexion of energy source and desalination plant.

Some situations and especially that of Tunisia will be examined.

Translation provided by courtesy of the submitting Government

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/825
ISRAEL
February 1964
Submitted in: ENGLISH and FRENCH

Confidential until official release during Conference

Abstract

NEUTRON COUNTING IN ACTIVATION ANALYSIS

S. Amiel

Israel Atomic Energy Commission

This paper reviews recent and current work done at this laboratory on the development and use of neutron counting in activation analysis. The use of neutrons for analysis, although not common practice, is very advantageous, since neutrons can be detected selectively and efficiently without interference from β and γ ray background. In addition, neutron emission is a relatively rare phenomenon in nuclear decay, and when taken together with other properties can be used for identification and assay of specific nuclides and elements. The same is true for the emission of neutrons from nuclear reactions with radioactive sources, where the emission is frequently characteristic of the source and target nuclides.

The specific features of emission and counting of neutrons permit a sample to be analysed without chemical processing prior to or after the irradiation, and facilitate rapid and non-destructive instrumental procedures at high precision, using relatively simple apparatus.

These neutron counting methods consist of the analysis of fissionable nuclides (e.g. U^{235}, U^{238}, U^{233}, Th^{232}, Pu^{239} and isotopic ratios of uranium) by counting the delayed neutron emission from short-lived fission products; analysis of Li^{6} and lithium, N^{15}, O^{17}, O^{18} and oxygen by counting delayed neutrons from 4 sec N^{17}; assay of several high energy gamma ray emitting nuclides (e.g. Na^{24}, Mn^{56}, La^{140} and Al^{28}) in highly radioactive samples where electronic discrimination is ineffective, by counting photoneutrons produced in D_2O and beryllium; and analysis of O^{18} in various phases by counting neutrons from the reaction $O^{18}(\alpha,n)Ne^{21}$.

The different methods and their applications are reported in detail. Further possibilities for extension and development of new methods along the same principles are discussed.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/826
ISRAEL
February 1964
Submitted in: ENGLISH and FRENCH

Confidential until official release during Conference

Abstract

THE THERMAL FLUX DISTRIBUTION IN THE MODERATOR REGION OF A SQUARE LATTICE OF CYLINDRICAL FUEL ELEMENTS

N. Margalith and Y. Yeivin

The Hebrew University, Jerusalem, Israel

The problem of deriving the mono-energetic flux distribution in a square lattice cell with a constant neutron source in the moderator region has been treated by several authors. Earlier studies neglected the absorption in the moderator, but more recently this has been taken into account. However, all these authors have only solved the problem in the diffusion approximation. Although it may be argued that the diffusion theory should hold in the moderator, this is certainly not true near the fuel rod. The exact flux shape near the fuel is important since it is mainly in this region that all extra absorbers, such as cans and coolant tubes, are situated. Also, one would like to know the true theoretical flux distribution, in order to be able to check and analyze flux measurements.

Correct flux distributions should, in principle, be calculated by one of the more or less sophisticated transport approximations. The purpose of the present investigation is to study a much simpler approach: one needs, essentially, only to solve the diffusion equation again; using the fuel disadvantage factor, F, and the extrapolation length at its surface, λ, this solution is easily corrected for transport effects.

The final solution is of the form

$$\Phi(x,y) = A[1 + \alpha_1 f(x,y; K_1) + \alpha_2 f(x,y; K_2)]$$

where f is the solution of the diffusion equation with the appro-

priate boundary conditions, K_1^2 is the reciprocal diffusion area, and K_2^2 is the second solution of the characteristic P_3 equation. The first two terms form the asymptotic (or diffusion) solution, and the third is the transport correction.

The coefficient α_1 is determined from λ, which is the reciprocal logarithmic derivative of the asymptotic flux. α_2 is determined by F, requiring flux continuity on the fuel-moderator boundary and the equality of the total absorption and source in the cell.

An ALTAC code has been written for a PHILCO-2000 computer, which gives the equi-flux curves in the moderator. These have been compared with a measured flux distribution in a natural uranium-heavy water lattice; the agreement is quite satisfactory.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/827
ISRAEL
February 1964
Submitted in: ENGLISH and FRENCH

Confidential until official release during Conference

Abstract

GENERALIZATION OF THE FERMI AGE THEORY AND ITS APPLICATION TO THE CALCULATION OF RESONANCE ESCAPE PROBABILITY

G. Rakavy and J.J. Wagschal

The Hebrew University, Jerusalem, Israel

A generalization of the Fermi Age Theory is obtained by replacing the expression $\xi \Sigma \varphi(u)$ - the slowing-down density - by the differential expression:

$$(1) \quad q(u) = \sigma_1(u)\varphi(u) + \frac{d}{du}[\sigma_2(u)\varphi(u)] + \frac{d^2}{du^2}[\sigma_3(u)\varphi(u)] + \ldots$$

The coefficients $\sigma_n(u)$ are defined by

$$(2) \quad \sigma_n(u) = \frac{1}{n!}\int v^n \sigma(u,v)dv$$

$\sigma(u,v)dv$ is the cross-section at lethargy u for a lethargy gain in the interval v, $v + dv$.

This expression for the slowing-down density q is obtained by expanding the collision density in the exact expression for q in a series and integrating term by term. The first term in (1) is Fermi's expression for the slowing-down density. Further terms describe the dispersion of neutrons in lethargy space during the slow-down.

The differential expression (1) for the slowing-down density may be applied in diffusion or transport calculations. We present results of resonance escape probability calculations obtained by truncating expression (1) after a few terms. The results are compared with other methods for calculating the resonance escape probability.

The method presented here has also been successfully applied to calculations of the slow-down of ions in a plasma.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/828
NETHERLANDS
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Choice of most suitable radiation characteristics for any given gauging-problem (Session F)

P. Platzek
Central Laboratory T.N.O., Delft-The Netherlands

Gauging mostly aims at detection or measurement of mass differences, rather than at determination of an average mass as such. In this respect level indication may be considered as a gauging problem as well, in addition to thickness- or density (difference!) measurements.

The problem can be simply presented as follows. A mass deviation ΔD from average mass D must be detected or measured by means of a signal difference ΔS significally larger than possible statistical signal fluctuations. This aim must be achieved in a given maximum time interval t. The available tools are isotope sources of different radiation characteristics such as mass attenuation coefficient and source strength.

When secondary effects as heterogeneity of radiation, scattering, source dimensions, detector response etc. are neglected, β- as well as γ-radiation can in first approximation be supposed to obey the same exponential absorption law. Simple theoretical considerations then show that the average object mass alone determines the choice of the optimum radio-isotope (c.q. mass attenuation coefficient of principal radiation). In consequence, the required source strength depends on

1. minimum mass difference to be detected (ΔD);
2. accuracy wanted ($\Delta S/\sqrt{S_D}$);
3. minimum measuring time t;
4. geometry and detector efficiency.

When signals are defined as total count = count rate x measuring time, formulae can be derived to show the intercorrelation of all important variables

and parameters in case the optimum radiation can be chosen. Moreover, a nomogram has been constructed to display the situation for optimum as well as for non-optimum conditions.

Experimental data show to which extent the simplified assumptions are valid. They justify the trial to consider all mass-difference measurements by means of high-energy radiation as a general problem, which can, in spite of the large variation in objective, always be approached in essentially the same way.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/829
UNITED ARAB REPUBLIC
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

DETERMINATION OF CERTAIN TRACE IMPURITIES IN PURE LEAD AND ALUMINIUM BY NEUTRON ACTIVATION ANALYSIS

H.K. El-Shamy, A.A. Abdel-Rassoul and T.Z. Bishay

The presence of microquantities of certain metallic impurities in lead generally limits its industrial application. Similarly, the presence of a number of foreign elements even as trace impurities in aluminium may also prevent its use as reactor material. The conventional methods of analysis for purity control of both metals are subjected to numerous limitations. Neutron activation analysis, however, has been commonly applied for ultra-micro quantitative determination of various impurities in a number of metals of high purity.

In the present work, a systematic determination of a considerable number of elements present as trace impurities in lead and aluminium samples of differing purity grades is described. A separation scheme based on co-precipitation and ion-exchange techniques is developed. The samples analysed are irradiated in the Inchass 2MW Research Reactor for differing periods with an activation neutron flux of $1.3 \times 10^{13} n/cm^2 s$ together with suitable standards of the elements to be determined. After 7 days' cooling, aliquot portions of the irradiated samples are dissolved in a suitable acid after the addition of representative carriers of the elements to be investigated. Both lead and aluminium are readily separable as chloride in excess HCl. The non-precipitable chlorides in concentrated hydrochloric acid are subsequently allowed to pass through a column of the strongly basic anionite, Dowex-1, and successfully fractionated with differing HCl/HF concentrations. The metals determined are mainly cobalt, iron, tin, zinc, cadmium, and chromium which have been further separated from the corresponding fractions by chemical precipitation. Moreover, the radiochemical purity of the radioactive nuclides concerned was checked before being compared with those induced

in the similarly irradiated respective standards. In this respect, the addition of carriers favours the possible purification of the radionuclides separated rather than their fractionation in carrier-free state.

The method proposed of successive ion-exchange separation and chemical precipitation thus ensures a complete decontamination from foreign activities. The procedures described can be used for routine analysis with good reproducibility.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/830
ITALY
February 1964
Submitted in:
ENGLISH and FRENCH

Confidential until official release during Conference

Abstract

CONTRIBUTION TO THE UNDERSTANDING OF THE NATURAL EVOLUTION OF PETROLEUM: RADIATION EFFECTS ON CRUDE OILS

U. Colombo and G. Sironi
Istituto G. Donegani, Novara, Italia
E. Denti and E. Zimmer
SORIN, Centro Ricerche Nucleari-Saluggia (Vercelli), Italia

In several regions of the world asphalt deposits associated with sedimentary radioactive rocks are known. It is believed that such accumulations, in some cases of considerable commercial importance, may have some connection with radiation effects on less complex molecules, such as liquid hydrocarbons, present in sedimentary basins.

Other agents (catalysis by minerals in contact rocks, oxidation, temperature, pressure) may have played a synergistic role with radiations in determining the chemical modification of hydrocarbons. Also the presence of bitumen with no commercial value - and yet significant when it comes to the exploration of oil fields - may be due to the radiation effects on simpler hydrocarbons migrating from their source rocks towards the reservoirs.

This paper reports the results of irradiation experiments on some Italian crude oils in the presence of different sedimentary rocks and water, in environmental conditions similar to those known for sedimentary basins. The total radiation doses employed are comparable to those which may be absorbed in geologic times by organic material associated with more or less radioactive shales.

Considerable radiation effects on the structure, distribution of molecular weights and viscosity of irradiated crudes were found.

Along with the formation of complex condensed molecules, light liquid and gaseous products are formed. Worthy of note the fact that irradiation in the presence of certain natural catalysts, particularly when water-wet, leads to the formation of much less hydrogen and unsaturated molecules than are obtained

when crude oils are irradiated without catalysts. This might explain why despite the primary role that radiation is believed to have played in the natural evolution of petroleum, at least in some instances neither hydrogen nor unsaturated products are very abundant in sedimentary basins.

In connection with these experiments it was found that irradiation of crude oil and of their distillation fractions in the presence of suitable catalysts brings about interesting chemical modifications which improve the quality of the oil in relation with its subsequent refining and chemical utilization.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/831
ITALY
February 1964
Submitted in: ENGLISH and FRENCH

Confidential until official release during Conference

Abstract

SPECIAL SAFEGUARD AND ERECTION FEATURES OF THE SELNI'S "ENRICO FERMI" NUCLEAR POWER PLANT

The design of the "Enrico Fermi" Nuclear Power Plant, SELNI, is presented and discussed from the safety standpoint.

The particular technical solutions adopted for taking into account the plant site characteristics and to increase the degree of safety typical of this type of plant.

Some special features of civil engineering, with particular reference to the problem of penetrations and to the various systems adopted to perform the big volume castings under particular conditions, the very thick floor form installation, the special linings with particular reference to the stainless steel lining for contaminable pools, etc. are also described.

The various steps of the electromechanical erections at the "Enrico Fermi" Nuclear Power Plant, SELNI, are also described. In particular, the field organization, the construction schedule and the description of the most important erection problems of a nuclear power plant as, for instance, lifting and location of heavy parts, welding performance and inspections thereof, seal and operating inspections, etc. are put in evidence.

The aim of the report is to bring the contribution of a direct experience, the construction and erection works having been performed directly by SELNI.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/832
GERMANY, FEDERAL REPUBLIC OF
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Fabrication Of Fused UO_2, UC Feed Material And Cast UC Rods.

P. Himmelstein, W. Hoppe, B. Liebmann, L. Schäfer, L. Schikarski and W. in der Schmitten

NUKEM, Wolfgang bei Hanau, Germany

Under sponsorship of the Joint US-EURATOM Program NUKEM has been carrying out development work on swaging and vibrational compaction of UO_2 and on the fabrication of UC feed material and cast rods. The first part of this paper describes a new technique for producing fused UO_2 suitable for vibrational compaction. In the second part research results on preparing stoichiometric UC and on casting UC in a centrifugal casting furnace are briefly summarized. A description of a production line for the manufacturing of 4.4 tons of cast UC rods for the EURATOM research centre at Ispra and the experience gained during the production are presented in more detail.

UO_2 is melted by passing an electric current through a column of loosely pressed UO_2 pellets stacked together between two electrodes. The furnace can be operated semicontinuously producing about 8 kg of fused UO_2 per batch with virtually no loss from the initial column of pellets. The crushed UO_2 is annealed in hydrogen to obtain an O/U ratio approximating 2.00. Analytical results show that the carbon content is below 100 ppm. The average nitrogen and hydrogen content is 50 and 10 ppm, respectively.

In UC development work stoichiometric feed material for melting was prepared by mixing UO_2 and graphite, pelletizing without a binder and reacting in a vacuum induction furnace at temperatures above 1600°C. The feed material was melted and cast into cylindrical rods in a centrifugal skull melting furnace. The carbon content was held to $\pm$ 0.1 wt% of the stoichiometric value. The average oxygen content amounted to 500 ppm.

The UC production consists of the following stages: Mixing of UO_2 and graphite, pelletizing, reacting to form the carbide, melting and casting, cutting and grinding, degreasing and annealing, dimensional control and packaging. The two main pieces of equipment used are a semicontinuous reaction furnace for the production of the feed material and an automatic tilt pour skull melting furnace. The production rate of the reaction furnace is 3 kg of UC per hour, and that of the melting furnace 50 kg of cast slugs in a day's shift. The carbon content can be kept within $\pm$ 0.1 wt% of the desired value. The average oxygen content is 500 ppm.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/833
CZECHOSLOVAKIA
February 1964
Submitted in:
ENGLISH and RUSSIAN

Confidential until official release during Conference

Abstract

THE METHOD OF MAIN PARAMETER DETERMINATION OF HEAVY WATER GAS COOLED POWER REACTOR CORE

J. Markvart, J. Pfann and P. Šulc

The presented paper describes the method and results of optimization calculations of heavy water power reactor core parameters. The method is based on the complex optimization of parameters from the point of view of main technical and economic indexes of an industrial power station.

Direct application of final optimization criteria concerning the whole power station in concrete applications restricts the choice of sufficient number of independent variable paramaters of reactors, primary and secondary circuits.

The analysis of general relations for useful power and net efficiency of the power plant leads to the determination of objective partial criteria for optimization of some active zone parameters. In many cases these criteria facilitate the judgement of specific problems connected with the design of the reactor.

It is possible in first approximation to use the criterion of the maximal effective fuel charge for optimization of the power reactor active zone arrangement. The effective fuel charge is expressed in the following:

$$U_{ef} = \frac{\Sigma \; u_i \; i}{\phi_{max}} \qquad \text{(summation over the whole volume of fuel)}$$

where: u_i is the section of reactor fuel irradiated by thermal neutron flux ϕ_i.

For the fixed heat rate of a fuel element irradiated by the maximum thermal neutron flux and determined heat removal conditions is the thermal and useful power of the reactor determined by the effective fuel charge.

The dependence of the effective fuel charge on geometrical parameters of the fuel element (the diameter and relative fuel element filling defined as ratio of uranium volume to fuel element volume) is searched for the one region core without reflector with various values of geometrical buckling of the assembly.

For practically available diameter range of fuel elements is the effective fuel charge only slightly variable. Therefore the variation of the fuel elements diameter can be adapted for design requirements or used to radial flattening of thermal neutron flux in a way as used in this paper.

For the specified type of fuel elements with determined fuel channel construction the effective fuel charge is increasing with increasing fuel element filling. But fuel element filling is one of main parameters which determine thermokinetical and hydrodynamical characteristics of the reactor, consequently it is necessary to determine it on the basis of the complex optimization. The results of these calculations for variable parameters as the gas flow, the heat exchanger outlet temperature of gas and the height of active zone enable to do important conclusions for the choice of the fuel element filling parameter.

The radial flattening of thermal neutron flux is effective means for increasing effective fuel charge. Two ways were followed: the flattening by means of fuel element having various diameters arranged into 3 regions and flattening caused by diminishing the moderator to fuel volume ratio in the central region when using press up elements and convenient combination of fuel elements having various diameters. For this case the possibility of peak xenon poisoning reactivity compensation by means of press up elements was assumed. The value of radial flattening coefficient 0.85 for optimum variant was reached.

Two group modified calculation method was used to determine critical dimensions and neutron flux distribution for multiregion core with multi-region reflector. Calculation of great number of variants was carried out using the small digital computer National Elliott 803 B. For this purpose the method was verified to be sufficiently accurate with satisfying time requirement for numerical solution. The computational program is expressed in Elliott-ALGOL language. In addition to possibility to determine critical dimensions and neutron flux distribution the program enables to compute reactivity changes for varying any of parameter in separate regions of the reactor.